UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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August 1, 2007

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SUBJECT: DIABLO CANYON POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000275/2007003 AND 05000323/2007003

Dear Mr. Keenan:

On June 30, 2007, the U.S. Nuclear Regulatory Commission completed an inspection at your Diablo Canyon Power Plant, Units 1 and 2, facility. The enclosed integrated report documents the inspection findings that were discussed on July 19, 2007, with Mr. James Becker and members of your staff.

This inspection examined activities conducted under your licenses as they relate to safety and compliance with the Commission's rules and regulations, and with the conditions of your licenses. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

There was one NRC-identified finding of very low safety significance (Green) identified in this report. This finding involved a violation of NRC requirements. Additionally, licensee identified violations which were determined to be of very low safety significance are listed in this report. However, because of their very low risk significance and because they are entered into your corrective action program, the NRC is treating these three findings as noncited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Diablo Canyon Power Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Vince G. Gaddy, Chief
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Division of Reactor Projects

Dockets: 50-275
50-323
Licenses: DPR-80
DPR-82

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and 05000323/2007003
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SUNSI Review Completed: __yes___ ADAMS: ■ Yes ☐ No Initials: __vgg___
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E-VGGaddy for	***E-VGGaddy for***	***/RA/***	***/RA/***
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Dockets:	50-275, 50-323
Licenses:	DPR-80, DPR-82
Report:	05000275/2007003 05000323/2007003
Licensee:	Pacific Gas and Electric Company
Facility:	Diablo Canyon Power Plant, Units 1 and 2
Location:	7 ½ miles NW of Avila Beach Avila Beach, California
Dates:	April 1 through June 30, 2007
Inspectors:	T. Jackson, Senior Resident Inspector M. Brown, Resident Inspector T. McKernon, Senior Operations Engineer, Operations Branch W. Sifre, Senior Reactor Inspector M. Hayes, Reactor Inspector G. Apger, Operations Engineer, Operations Branch D. Stearns, Health Physicist, Plant Support Branch
Approved By:	V. G. Gaddy, Chief, Projects Branch B Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000275/2007-003, 05000323/2007-003; 4/1/07 - 6/30/07; Diablo Canyon Power Plant Units 1 and 2; Maintenance Effectiveness.

This report covered a 13-week period of inspection by resident inspectors and announced inspections in occupational radiation protection, licensed operator requalification, and inservice inspection activities. One NRC-identified, Green, noncited violation was identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609 “Significance Determination Process.” Findings for which the Significance Determination Process does not apply may be Green or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR 50.65(b) for the failure of engineering personnel to include the reactor cavity and containment structure sump level indication systems into the scope of its program for monitoring the effectiveness of maintenance. Specifically, between April 14, 2007, and May 17, 2007, Units 1 and 2 experienced multiple failures of the reactor cavity and containment structure sump level indications. These systems are required by the plant’s Technical Specifications in order to promptly identify and take actions for reactor coolant system leaks before they can potentially develop into a loss of coolant accident. Additionally, the inspectors discovered that Emergency Operating Procedure ECA-3.1, “SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired,” Revision 18, utilized the containment structure sump level indication for mitigative actions. Based on the fact that the systems are used to mitigate a loss of coolant accident and were used in the emergency operating procedures, the inspectors determined that the systems should have been included in Pacific Gas and Electric Company’s program for monitoring the effectiveness of maintenance. This issue was entered into Pacific Gas and Electric Company’s corrective action program as Action Request A0696295.

  The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Using Inspection Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to have very low safety significance since it did not represent a loss of system safety function, an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, or screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. This finding has a crosscutting aspect in the area of human performance, associated with the decision-making component, in that Pacific Gas and Electric Company failed to use conservative assumptions in evaluating the function and use of the sump level indications in mitigating the effects of design basis accidents (H.1(b)) (Section 1R12).

B. Licensee-Identified Violations

Violations of very low safety significance, which have been identified by Pacific Gas and Electric Company, have been reviewed by the inspectors. Corrective actions taken or planned by Pacific Gas and Electric Company have been entered into their corrective action program. These violations and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Diablo Canyon Unit 1 began this inspection period at 100 percent power and entered Refueling Outage 1R14 on April 30, 2007. Unit 1 entered Mode 6 (Refueling) for core offload operations on May 4, which was completed on May 7. Unit 1 entered Mode 6 on May 16, when operators began reloading fuel into the core, and then entered Mode 5 (Cold Shutdown) on May 22 when maintenance personnel tensioned the reactor vessel head. Operators commenced a heatup of the reactor coolant system (RCS), and Unit 1 entered Mode 4 (Hot Shutdown) on May 25 and Mode 3 (Hot Standby) on May 26. On May 28, operators proceeded with reactor startup, entering Mode 2 (Startup). Operators increased reactor power, and Unit 1 entered Mode 1 (Power Operations) on May 28. On May 29, Unit 1 was paralleled to the grid, ending Refueling Outage 1R14. Operators continued to raise reactor power and, on June 2, Unit 1 reached 100 percent power and remained at that power level for the remainder of the inspection period.

Diablo Canyon Unit 2 operated at 100 percent power for the duration of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignments (71111.04)

Partial System Walkdowns

a. Inspection Scope

The inspectors: (1) walked down portions of the three below listed risk-important systems and reviewed plant procedures and documents to verify that critical portions of the selected systems were correctly aligned; and (2) compared deficiencies identified during the walk down to the Final Safety Analysis Report (FSAR) Update and the corrective action program (CAP) to ensure problems were being identified and corrected.

- April 3, 2007: Unit 1, Residual Heat Removal Pump 1-1
- April 10, 2007: Unit 1, Auxiliary Saltwater Pump 1-1
- May 20, 2007: Unit 1, Reactor Vessel

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Inspection

a. Inspection Scope

The inspectors walked down the six below listed plant areas to assess the material condition of active and passive fire protection features and their operational lineup and readiness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify that they remained functional; (3) observed fire suppression systems to verify that they remained functional and that access to manual actuators was unobstructed; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features and that the compensatory measures were commensurate with the significance of the deficiency; and (7) reviewed the FSAR Update to determine if Pacific Gas and Electric Company (PG&E) identified and corrected fire protection problems.

- April 3, 2007, Unit 1, Residual heat removal pump rooms
- May 8, 2007, Unit 1, Containment 91 ft. level
- May 14, 2007, Unit 1, Battery rooms
- June 5, 2007, Unit 1, Main lube oil cooler room
- June 20, 2007, Unit 2, Battery rooms
- June 21, 2007, Unit 1, Component cooling water and containment spray pump rooms

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Performance of Nondestructive Examination (NDE) Activities Other than Steam Generator Tube Inspections, Pressurized Water Reactor (PWR) Vessel Upper Head Penetration Inspections, Boric Acid Corrosion Control

a. Inspection Scope

The inspection procedure required the review of NDE activities consisting of two or three different types (i.e., volumetric, surface, or visual). The inspectors observed the performance of one liquid penetrant examination (surface), one radiographic examination (volumetric), and two visual examinations. The inspectors also reviewed four ultrasonic examinations.

For each of the observed NDE activities, the inspectors verified that the examinations were performed in accordance with the specific site procedures and the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements.

During the review of each examination, the inspectors verified that appropriate NDE procedures were used, examinations and conditions were as specified in the procedure, and test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspectors also verified the NDE certifications of the personnel who performed the above examinations. Finally, the inspectors verified that the indications identified during the examinations were dispositioned in accordance with the ASME Code-qualified NDE procedures used to perform the examinations.

The inspection procedure required review of one or two examinations with recordable indications that were accepted for continued service to ensure that the disposition was made in accordance with the ASME Code. PG&E did not accept any examinations with recordable indications for continued service.

The inspection procedure further required verification of one to three welds on Class 1 or 2 pressure boundary piping to ensure that the welding process and welding examinations were performed in accordance with the ASME Code. The inspectors reviewed two welds performed on the component cooling water system. The inspectors verified that the welding was performed in accordance with the ASME Code. This included the review of welding material issue slips to establish that the appropriate welding materials had been used and the verification of the welding procedure specifications had been properly qualified.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Reactor Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

The inspection requirements for this section parallel the inspection requirement steps in Section 1R08.1 listed above. PG&E performed examinations of all 79 control rod drive mechanism penetration nozzles and one vent line. The examination methods used varied depending on the penetration tube configuration. All of the penetration tubes were examined using time-of-flight diffraction ultrasonic examination combined with zero-degree straight beam examination to identify evidence of a leak path in the shrink-fit area. The examinations were supplemented with eddy current examinations.

The inspectors reviewed the examination procedures used and confirmed that the equipment and calibration requirements (essential variables) were consistent with that used in vendor mockup demonstrations. The inspectors reviewed the records recording the extent of inspection for each penetration nozzle including the resolution of interference and masking issues. The inspectors verified that the activities were performed in accordance with the requirements of NRC Order EA-03-009. There were no detectible defects identified and no weld repairs performed.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control Inspection Activities (PWRs)

a. Inspection Scope

The inspectors evaluated the implementation of PG&E's boric acid corrosion control program for monitoring degradation of those systems that could be adversely affected by boric acid corrosion. The inspection procedure required review of a sample of boric acid corrosion control walkdown visual examination activities through either direct observation or record review. The inspectors reviewed the documentation associated with PG&E's boric acid corrosion control walkdown.

The inspection procedure requires verification that visual inspections emphasize locations where boric acid leaks can cause degradation of safety significant components. The inspectors verified through direct observation and program/record review that PG&E's boric acid corrosion control inspection efforts were directed towards locations where boric acid leaks could cause degradation of safety-related components.

The inspection procedure required both a review of one to three engineering evaluations performed for boric acid leaks found on reactor coolant system (RCS) piping and components and one to three corrective actions performed for identified boric acid leaks. The inspectors reviewed two evaluations to assess PG&E's analysis and evaluate the assessment of the condition and proposed corrective actions.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspection procedure specified performance of an assessment of in-situ screening criteria to assure consistency between assumed nondestructive examination flaw sizing accuracy and data from the Electric Power Research Institute (EPRI) examination technique specification sheets. It further specified assessment of appropriateness of tubes selected for in-situ pressure testing, observation of in-situ pressure testing, and review of in-situ pressure test results.

No conditions were identified that warranted in-situ pressure testing. The inspectors did, however, review PG&E's degradation assessment report, "Steam Generator Degradation Assessment for 1R14," Revision 0, and compared the in-situ test screening parameters to the EPRI guidelines. This review determined that the screening parameters were consistent with the EPRI guidelines.

In addition, the inspectors reviewed both PG&E's site-validated and qualified acquisition and analysis technique sheets used during this refueling outage and the qualifying EPRI examination technique specification sheets to verify that the essential variables regarding flaw sizing accuracy, tubing, equipment, technique, and analysis had been identified and qualified through demonstration.

The inspection procedure specified comparing the estimated size and number of tube flaws detected during the current outage against the previous outage operational assessment predictions to assess PG&E's prediction capability. The inspectors compared the previous outage operational assessment predictions with the flaws identified during the current steam generator tube inspection effort. Compared to the projected damage mechanisms identified by PG&E, the number of identified indications fell within the range of prediction and were consistent with those predictions.

The inspection procedure specified confirmation that the steam generator tube eddy current test scope and expansion criteria meet Technical Specification requirements, EPRI guidelines, and commitments made to the NRC. The inspectors evaluated the recommended steam generator tube eddy current test scope established by Technical Specification (TS) requirements and the degradation assessment report. The inspectors compared the recommended test scope to the actual test scope and found that PG&E had accounted for all known flaws and had, as a minimum, established a test scope that met TS requirements, EPRI guidelines, and commitments made to the NRC.

The inspection procedure specified, if new degradation mechanisms were identified, verification that PG&E fully enveloped the problem in its analysis of extended conditions including operating concerns and had taken appropriate corrective actions before plant startup. No new degradation mechanisms were identified.

The inspection procedure required confirmation that PG&E inspected all areas of potential degradation, especially areas that were known to represent potential eddy current test challenges (e.g., top-of-tubesheet and tube support plates). The inspectors confirmed that all known areas of potential degradation were included in the scope of inspection and were being inspected.

The inspection procedure also required confirmation of adherence to the TS plugging limit, unless alternate repair criteria have been approved. The inspection procedure further required determination whether depth sizing repair criteria were being applied for indications other than wear or axial primary water stress corrosion cracking in dented tube support plate intersections. The inspectors determined that the TS plugging limits were being adhered to (i.e., 40 percent maximum through-wall indication).

If steam generator leakage greater than three gallons per day was identified during operations or during post shutdown visual inspections of the tubesheet face, the inspection procedure required verification that PG&E had identified a reasonable cause based on inspection results and that corrective actions were taken or planned to address the cause for the leakage. The inspectors did not conduct any assessments because this condition did not exist.

The inspection procedure required confirmation that the eddy current test probes and equipment were qualified for the expected types of tube degradation and an assessment of the site-specific qualification of one or more techniques. The inspectors reviewed portions of eddy current tests performed on the tubes in all four steam generators. The inspectors verified that: (1) the probes appropriate for identifying the expected types of indications were being used, (2) probe position location verification was performed, (3) calibration requirements were adhered to, and (4) probe travel speed was in accordance with procedural requirements. The inspectors performed a review of site-specific qualifications of the techniques being used.

If loose parts or foreign material on the secondary side were identified, the inspection procedure specified confirmation that PG&E had taken or planned appropriate repairs of affected steam generator tubes and that they inspected the secondary side to either remove the accessible foreign objects or perform an evaluation of the potential effects of inaccessible object migration and tube fretting damage. No loose parts or foreign material were identified.

Finally, the inspection procedure specified review of one to five samples of eddy current test data if questions arose regarding the adequacy of eddy current test data analyses. The inspectors did not identify any results where the adequacy of eddy current test data analysis was questionable.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspection procedure required review of a sample of problems associated with inservice inspections documented by PG&E in the CAP for appropriateness of the corrective actions. For this sample, the inspectors reviewed three action requests which dealt with inservice inspection and welding activities. From this review, the inspectors concluded that PG&E has an appropriate threshold for entering issues into the corrective action program and has procedures that direct a root cause evaluation when necessary. PG&E also had an effective program for applying industry operating experience.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

.1 Quarterly Inspection

a. Inspection Scope

On June 12, 2007, the inspectors observed testing and training of senior reactor operators and reactor operators to identify deficiencies and discrepancies in training, to assess operator performance, and to assess the evaluator's critique. The training scenario involved a nuclear instrument failure, main feedwater pump high vibration, a faulted and ruptured steam generator, and an anticipated transient without scram.

Documents reviewed by the inspectors included Lesson ES1213-B, "LOCA," Revision 12.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Biennial Inspection

a. Inspection Scope

The inspectors: (1) evaluated examination security measures and procedures in compliance with 10 CFR 55.49; (2) evaluated PG&E's sample plan for the written examinations in compliance with 10 CFR 55.59 and NUREG-1021, “Operators Licensing Examination Standards for Power Reactor, Revision 9," as referenced in the facility requalification program procedures; and (3) evaluated maintenance of license conditions in compliance with 10 CFR 55.53 by reviewing the facility records (medical and administration), procedures, and tracking systems for licensed operator training, qualification, and watchstanding. In addition, the inspectors reviewed remediation training and examinations for examination failures in compliance with facility procedures and responsiveness to address areas failed. The inspectors also verified that on-shift operators requiring prescription lenses for self-contained breathing apparatus maintained their lenses secured in the control room.

Furthermore, the inspectors: (1) interviewed six personnel (one operator, two senior operators, two instructors/evaluators, and a training supervisor) regarding the policies and practices for administering examinations; (2) observed the administration of two dynamic simulator scenarios to three requalification crews (two shift crews and one administrative crew) by facility evaluators, including an operations department manager, who participated in the crew and individual evaluations; and (3) observed three facility evaluators administer five job performance measures, including two in the control room simulator in a dynamic mode, and three in the plant under simulated conditions. Each job performance measure was observed by at least two requalification candidates.

The inspectors also reviewed the biennial written examinations including two remediation written examinations for a reactor operator and a senior reactor operator. The inspectors verified/questioned the level of difficulty, knowledge level, and overlap between successive examinations and remedial examinations. In addition, quality audits and training self-assessments, and training management meeting minutes were reviewed to ascertain the health of their training feedback processes.

Of the 70 licensed operators taking the biennial examinations, one shift crew and one administrative crew failed the dynamic simulator scenario portion of the examination. Both crews were remediated, retested and passed the remediation examination. In addition, one reactor operator and one senior reactor operator failed the written examination. Both individuals were remediated, retested and passed the remediation examination. The inspectors also reviewed the results of the annual licensed operator requalification operating examinations for 2006 and 2007. The results of the examinations were also reviewed to assess PG&E’s appraisal of operator performance and the feedback of that performance analysis to the requalification training program. The inspectors also observed the examination security maintenance during the examination week.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Maintenance Effectiveness Inspection

a. Inspection Scope

The inspectors reviewed the three below listed maintenance activities to: (1) verify the appropriate handling of structure, system, and component (SSC) performance or condition problems; (2) verify the appropriate handling of degraded SSC functional performance; (3) evaluate the role of work practices and common cause problems; and (4) evaluate the handling of SSC issues reviewed under the requirements of the Maintenance Rule, 10 CFR Part 50, Appendix B, and the TSs.

- April 9, 2007, Unit 2, Inter-system loss-of-coolant accident boundary valves
- April 16, 2007, Units 1 and 2, Failure of Reactor Cavity Sump Level Indication
- April 17, 2007, Unit 2, Residual Heat Removal Check Valve RHR-2-8740A

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

Introduction. An NRC identified, Green, noncited violation (NCV) of 10 CFR 50.65(b) was identified for the failure of engineering personnel to include the reactor cavity and containment structure sump level indication systems into the scope of its program for monitoring the effectiveness of maintenance. Specifically, between April 14, 2007, and May 17, 2007, Units 1 and 2 experienced multiple failures of the reactor cavity and containment structure sump level indications. These systems are required by the plant's TS in order to promptly identify and take actions for reactor coolant system (RCS) leaks before they can potentially develop into a loss of coolant accident. Additionally, the inspectors discovered that Emergency Operating Procedure (EOP) ECA-3.1, "SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired," Revision 18, utilized the containment structure sump level indication for mitigative actions. Based on the fact that the systems are used to mitigate a loss of coolant accident and were used in the EOPs, the inspectors determined that the systems should have been included within the scope of PG&E's program for monitoring the effectiveness of maintenance.

Description. Beginning on April 17, 2007, the inspectors observed several occasions where the reactor cavity sump level indication for both units behaved erratically. Suspected sources of the failures included a loose indicator faceplate and bubbler tube blockage. The Unit 1 reactor cavity sump level indication was declared inoperable when the unit entered Refueling Outage 1R14. After a blowdown of the bubbler tubes on both units' reactor cavity sump level indication systems, they were returned to service. The containment structure sump level indicators (LI-60 and LI-61) have also had several failures since 2005. The failures were due to various causes, including stuck gage needles, and binding of the needle with the gage scale. The cause of one failure of Unit 1 LI-60 was not determined and the instrument was replaced.

The inspectors questioned engineering personnel on whether the reactor cavity and containment structure sump level indications were included in their program for monitoring the effectiveness of maintenance on plant SSCs. The inspectors found that these systems are part of the liquid radwaste system, which was not scoped into the maintenance rule. According to the FSAR Update, Section 3.6.2.1.1.1, "the leak-before-break analysis also assumes that the DCPP reactor coolant system leak detection system has the capability to detect an increase in reactor coolant system leakage into the containment of 1 gpm. The current design basis for this system indicates that it has this capability. Operability of this system is controlled by the plant Technical Specifications [TS 3.4.15]." Therefore, the inspectors concluded that both the containment structure sumps and reactor cavity sump level indications are relied upon to mitigate the effects of an FSAR-described accident (i.e., loss of coolant accident). Additionally, the inspectors discovered that the containment structure sump level indication was included in EOP ECA-3.1, "SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired," Revision 18. Based on its use in an EOP, and its use to mitigate an FSAR-described accident by providing an RCS leak-before-break indication, the inspectors concluded that the containment structure and reactor cavity sump level indications should have been included within the scope of PG&E's program for monitoring the effectiveness of maintenance on plant SSCs.

PG&E is continuing to monitor the containment structure and reactor cavity sump level indications and troubleshoot the cause of the failures.

<u>Analysis</u>. The performance deficiency associated with this finding was the failure of engineering personnel to properly scope the systems associated with reactor cavity and containment structure sump level indication. The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance since it did not represent a loss of system safety function, an actual loss of safety function of a single train for greater than its TS allowed outage time, or screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. This finding has a crosscutting aspect in the area of human performance, associated with the decision-making component, in that PG&E failed to use conservative assumptions in evaluating the function and use of the sump level indications in mitigating the effects of design basis accidents (H.1(b)).

<u>Enforcement</u>. 10 CFR 50.65(b) requires, in part, that the scope of the monitoring program specified in paragraph (a)(1) of this section shall include nonsafety related structures, systems, or components that are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures . Contrary to this, engineering personnel failed to properly scope the necessary structures, systems, and components associated with reactor cavity and containment structure sump level

indication into the PG&E maintenance monitoring program. Specifically, the inspectors observed the containment structure sump level indication being used in EOP ECA-3.1, and also observed that the level indications were credited in the FSAR Update and TS for providing prompt identification and actions to avoid a potential loss-of-coolant accident in the event of an RCS leak. Because the finding is of very low risk significance and has been entered into the CAP as AR A0696295, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 05000275/2007003-01, "Failure to Scope Reactor Cavity and Containment Structure Sumps Level Indication Into Maintenance Rule."

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

.1 Risk Assessments and Management of Risk

a. Inspection Scope

The inspectors reviewed the four below listed assessment activities to verify: (1) performance of risk assessments when required by 10 CFR 50.65(a)(4) and PG&E procedures prior to changes in plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information considered in the risk assessment; (3) that PG&E recognizes, and/or enters as applicable, the appropriate risk category according to the risk assessment results and PG&E procedures; and (4) PG&E identified and corrected problems related to maintenance risk assessments.

- April 4, 2007, Unit 1, Scheduled maintenance for Component Cooling Water Pump 1-3, Eagle 21 Rack 11 software, Diablo-Gates 500 kV line, and Morro Bay-Mesa 230 kV line

- April 6, 2007, Unit 1, Positive displacement pump replacement

- April 9, 2007, Unit 1, 4 kV Bus G cubicle SGH11 maintenance

- May 15, 2007, Unit 1, Transfer of single source of offsite power from 230 kV to 500 kV during refueling outage

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plant status documents such as operator shift logs, emergent work documentation, deferred modifications, and standing orders to determine if an operability evaluation was warranted for degraded components; (2) referred to the FSAR Update and design bases documents to review the technical adequacy of the operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any TS; (5) used the Significance Determination Process to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that PG&E has identified and implemented appropriate corrective actions associated with degraded components.

- April 3, 2007, Unit 1, Component cooling water return Header A Pipe Support 55S-180R alignment
- April 16, 2007, Units 1 and 2, Operating with Tave less than design value
- April 10, 2007, Units 1 and 2, Cavitation erosion downstream of auxiliary feedwater recirculation line reducing orifice
- April 20, 2007, Unit 2, Diesel Engine Generator 2-3 jacket water pump leakage
- May 21, 2007, Unit 1, Diesel Engine Generator 1-3 lube oil leak and broken bolt on starting air motor mount

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the six below listed postmaintenance test activities of risk-significant systems or components. For each item, the inspectors: (1) reviewed the applicable licensing basis and/or design basis documents to determine the safety functions; (2) evaluate the safety functions that may have been affected by the maintenance activity; and (3) reviewed the test procedure to ensure it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, the test data results were complete and accurate, the test equipment was removed, the system was properly realigned, and deficiencies during testing were documented. The inspectors also reviewed the FSAR Update to determine if PG&E identified and corrected problems related to post-maintenance testing.

- April 2, 2007, Unit 1, Main feedwater bypass Valve FCV-1540 linear variable differential transformer replacement
- April 14, 2007, Units 1 and 2, Reactor cavity sump level Indication LI-62 erratic indication
- April 24, 2007, Unit 2, Diesel Engine Generator 2-3 jacket water pump replacement
- June 4, 2007, Unit 1, Battery 1-1 Cell 15 replacement
- June 5, 2007, Unit 1, Moveable incore detection system thimble tube replacements
- June 18, 2007, Unit 1, Digital feedwater control system installation

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the following risk-significant refueling items or outage activities to verify defense-in-depth commensurate with the outage risk control plan, compliance with the TS, and adherence to commitments in response to Generic Letter 88-17, "Loss of Decay Heat Removal": (1) the risk control plan; (2) tagging/clearance activities; (3) RCS instrumentation; (4) electrical power; (5) decay heat removal; (6) spent fuel pool cooling; (7) inventory control; (8) reactivity control; (9) containment closure; (10) reduced inventory or midloop conditions; (11) refueling activities; (12) heatup and cooldown activities; (13) restart activities; (14) identification and implementation of appropriate corrective actions associated with refueling and outage activities. The inspectors' containment inspections included observations of the containment sump for damage and debris and supports, braces, and snubbers for evidence of excessive stress, water hammer, or aging. Documents reviewed by the inspectors included the Unit 1 Refueling Outage 1R14 Outage Safety Plan.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the FSAR Update, procedure requirements, and TS to ensure that the five below listed surveillance activities demonstrated that the SSCs tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the following significant surveillance test attributes were adequate: (1) preconditioning; (2) evaluation of testing impact on the plant; (3) acceptance criteria; (4) test equipment; (5) procedures; (6) jumpers; (7) test data; (8) testing frequency and method demonstrated TS operability; (9) test equipment removal; (10) restoration of plant systems; (11) fulfillment of American Society of Mechanical Engineers (ASME) Code requirements; (12) updating of performance indicator data; (13) engineering evaluations, root causes, and bases for returning tested SSCs not meeting the test acceptance criteria were correct; (14) reference setting data; and (15) annunciators and alarm setpoints. The inspectors also verified that PG&E identified and implemented any needed corrective actions associated with the surveillance testing.

- April 2, 2007, Unit 2, Inservice inspection of mechanical snubbers
- April 5, 2007, Unit 1, Comprehensive inservice testing of Auxiliary Feedwater Pump 1-1 (Pump Inservice Test)
- April 12, 2007, Unit 2, Reactor coolant pressure boundary leakage monitoring program (RCS leak detection testing)
- May 20, 2007, Unit 1, Integrated test of engineered safeguards and diesel generators
- June 13, 2007, Unit 1, Containment isolation valve leak testing (Containment Isolation Valve Testing)

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample of a pump inservice test, one sample of a containment isolation valve test, one sample of a RCS leak detection test, and two other surveillance tests for a total of five samples.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Emergency Preparedness Evaluation (71114.06)

a. Inspection Scope

For the one below listed drill contributing to Drill/Exercise Performance and Emergency Response Organization Performance Indicators, the inspectors: (1) observed the training evolution to identify any weaknesses and deficiencies in the emergency response organization; (2) compared the identified weaknesses and deficiencies against PG&E identified findings to determine whether PG&E is properly identifying failures; and (3) determined whether PG&E performance is in accordance with the guidance of the NEI 99-02, "Voluntary Submission of Performance Indicator Data," acceptance criteria.

- June 8, 2007, Units 1 and 2, Rapid response drill for the emergency response organizations

Documents reviewed by the inspectors included the Diablo Canyon Power Plant Emergency Plan, Revision 4.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control To Radiologically Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess PG&E's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. The inspectors used the requirements in 10 CFR Part 20, the TSs, and PG&E's procedures required by TSs as criteria for determining the compliance. During the inspection, the inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation packages reported by PG&E in the Occupational Radiation Safety Cornerstone

- Controls (surveys, posting, and barricades) of three radiation, high radiation, or airborne radioactivity areas

- Radiation work permits, procedures, engineering controls, and air sampler locations

- Conformity of electronic personal dosimeter alarm set points with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms

- Physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools

- Self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection

- Corrective action documents related to access controls

- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies

- Radiation work permit briefings and worker instructions

- Adequacy of radiological controls, such as required surveys, radiation protection job coverage, and contamination control during job performance

- Dosimetry placement in high radiation work areas with significant dose rate gradients

- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas

- Controls for special areas that have the potential to become very high radiation areas during certain plant operations

- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas

- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

The inspectors completed 19 samples.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors assessed PG&E's performance in regards to maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspectors used the requirements in 10 CFR Part 20 and PG&E's procedures required by the TSs as criteria for determining compliance. The inspectors interviewed PG&E personnel and reviewed:

- Five outage or online maintenance work activities scheduled during the inspection period and associated work activity exposure estimates which were likely to result in the highest personnel collective exposures

- Site-specific trends in collective exposures, plant historical data, and source-term measurements

- Site-specific ALARA procedures

- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements

- Interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling and engineering groups

- Integration of ALARA requirements into work procedure and radiation work permit (or radiation exposure permit) documents

- Shielding requests and dose/benefit analyses

- Dose rate reduction activities in work planning

- Exposure tracking system

- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding

- Workers' use of the low dose waiting areas

- Records detailing the historical trends and current status of tracked plant source terms and contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry

- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas

- Self-assessments, audits, and special reports related to the ALARA program since the last inspection

- Resolution through the corrective action process of problems identified through post-job reviews and post-outage ALARA report critiques

- Corrective action documents related to the ALARA program and followup activities, such as initial problem identification, characterization, and tracking

- Effectiveness of self-assessment activities with respect to identifying and addressing repetitive deficiencies or significant individual deficiencies

The inspectors completed 17 samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

.1 Mitigating Systems Cornerstone

a. Inspection Scope

The inspectors sampled PG&E submittals for the PIs listed below for the period of July 2006 to June 2007, for Units 1 and 2. The definitions and guidance of NEI 99-02, “Regulatory Assessment Indicator Guideline,” Revision 4, were used to verify PG&E’s basis for reporting each data element in order to verify the accuracy of PI data reported during the assessment period. The inspectors reviewed licensee event reports, monthly operating reports, and operating logs as part of the assessment.

- Safety System Functional Failures
- Emergency AC Power System
- High Pressure Safety Injection System
- Auxiliary Feedwater System
- Residual Heat Removal System
- Cooling Water Support System

The inspectors completed six samples per unit.

b. Findings

No findings of significance were identified.

.2 Occupational Radiation Safety Cornerstone

a. Inspection Scope

The inspectors reviewed the one below PI from July 1, 2006, through March 31, 2007. The review included corrective action documentation that identified occurrences in locked high radiation areas (as defined in PG&E’s TSs), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 4). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspectors interviewed PG&E personnel that were accountable for collecting and evaluating the performance indicator data. In addition, the inspectors toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. Performance indicator definitions and guidance contained in NEI 99-02, Revision 4, were used to verify the basis in reporting for each data element.

- Occupational Exposure Control Effectiveness

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.3 Public Radiation Safety Cornerstone

a. Inspection Scope

The inspectors reviewed the one below PI from July 1, 2006, through March 31, 2007. PG&E records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded performance indicator thresholds and those reported to NRC. The inspectors interviewed PG&E personnel who were accountable for collecting and evaluating the performance indicator data. Performance indicator definitions and guidance contained in NEI 99-02, Revision 4, were used to verify the basis in reporting for each data element.

- Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a daily screening of items entered into PG&E's CAP. This assessment was accomplished by reviewing ARs and event trend reports, and attending daily operational meetings. The inspectors: (1) verified that equipment, human performance, and program issues were being identified by PG&E at an appropriate threshold and that the issues were entered into the corrective action program; (2) verified that corrective actions were commensurate with the significance of the issue; and (3) identified conditions that might warrant additional follow-up through other baseline inspection procedures.

b. Findings

No findings of significance were identified.

.2 Selected Issue Follow-Up Inspection

a. Inspection Scope

In addition to the routine review, the inspectors selected the one below listed issue for a more in-depth review. The inspectors considered the following during the review of PG&E's actions: (1) complete and accurate identification of the problem in a timely manner; (2) evaluation and disposition of operability/reportability issues; (3) consideration of extent of condition, generic implications, common cause, and previous occurrences; (4) classification and prioritization of the resolution of the problem; (5) identification of root and contributing causes of the problem; (6) identification of corrective actions; and (7) completion of corrective actions in a timely manner.

- May 26, 2007, Unit 1, Accumulator Voiding

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.3 Semiannual Trend Review

a. Inspection Scope

The inspectors completed a semi-annual trend review of repetitive or closely related issues that were documented in action requests, maintenance rule reports, system health reports, problem lists, and performance indicators to identify trends that might indicate the existence of more safety significant issues. The inspectors review consisted of the six-month period from January to June 2007. When warranted, some of the samples expanded beyond those dates to fully assess the issue. Corrective actions associated with a sample of the issues identified in PG&E's trend report were reviewed for adequacy. Documents reviewed by the inspectors are listed in the attachment.

b. Findings

During the review period from January to June 2007, the inspectors noted several instances of corrosion associated with safety-related structures, systems, and components. Specifically, the inspectors noted corrosion issues associated with the containment fan cooler units (CFCUs), the control room ventilation system, and the intake structure.

CFCUs

Each containment building at the Diablo Canyon Power Plant includes five CFCUs. The CFCUs are safety-related and relied upon to remove containment heat, and thus reduce the containment pressure, following a design bases accident. Each CFCU contains two banks of cooling coils, with each bank consisting of six coil assemblies stacked one on top of each other. The structural support for the coil assemblies is provided by the steel brackets on each end of the assemblies, and the brackets are bolted to the CFCU outer frame. Between each coil assembly within a bank is a separator. The separator is manufactured of galvanized sheet metal, and it appears that its purpose is to prevent bypass air flow between the coil assemblies within the bank.

PG&E maintenance and engineering personnel have noted corrosion of the coil assembly separators since 1998. In the recent Unit 1 Refueling Outage 1R14, PG&E personnel and the inspectors observed severe through-wall corrosion of the separators on at least two CFCUs and to a lesser extent, there was corrosion of separators in other CFCUs. The inspectors reviewed the impact on CFCU operability that the corrosion may have and determined that currently there was no impact. Specifically, the separators did not provide any structural support in the CFCUs. The inspectors also noted that there were no areas for the air to bypass the cooling coils through the separators. The corrosion products do have the potential to impact the functionality of the CFCU drain pan level instrumentation. The CFCU drain pan level instrumentation is one of three methods that are used to identify reactor coolant system leakage. In the past, the corrosion products had impacted CFCU drain pan level instrumentation. However, per PG&E's maintenance monitoring program, PG&E personnel increased the frequency of the CFCU drain pan level instrumentation flush from every third operating cycle to every operating cycle. The increase in flushes appeared to be successful since there were no additional issues with the drain pan level instrumentation since the preventive maintenance change.

While the inspectors determined that there were no current operability issues with the corrosion in the CFCUs, future corrosion rates are expected to be faster and the impact to be larger. Specifically, the corrosion of the separators may allow bypass air flow around the cooling coils or generate sufficient corrosion products to impact the operation of the CFCU drain pan level instrumentation prior to its preventive maintenance in the refueling outages. Engineering personnel currently plan to have the CFCU cooling coils and separators replaced in the next five to six years.

<u>Control Room Ventilation System</u>

NRC Inspection Report 05000275; 323/2007002 documented a finding related to the Unit 2 Control Room Condenser CR-38. In August 2006, while performing paint preparations for the control room condenser, maintenance personnel discovered large amounts of through-wall corrosion on the condenser's filter housing. During the process of corrosion removal, at least two of the support bars on the filter housing were broken. Some areas of the through-wall corrosion were approximately 16 inches^2. As a result of the corrosion, the operators declared the control room condenser inoperable due to the inability to determine seismic qualification. PG&E has planned to replace the filter housing in the next maintenance outage window for Control Room Condenser CR-38.

<u>Intake Structure</u>

In March 2006, PG&E placed the intake structure into the maintenance rule (a)(1) goal setting, due to an observed adverse trend in corrosion and concrete degradation. This is the second time that the intake structure has been placed in (a)(1) status. Between January and June 2007, condition reports were written by PG&E identifying additional areas of saltwater intrusion and concrete degradation, including broken concrete in the ceiling near Hatches 22 and 23 (AR A0688493) and saltwater intrusion in the ceiling west of Circulating Water Pump 1-2 (AR A0693877). Additionally, repairs were made to the Unit 1 Auxiliary Saltwater Pump vaults to correct some known degradation, but additional repairs were deferred until the next Unit 1 refueling outage (AR A0682505 and A0695032).

While the inspectors determined that there were no current operability issues with the corrosion in the intake structure, the inspectors concurred with engineering personnel that the continued adverse trend in degradation could result in the intake structure losing its design margin and violating its design basis criteria. Engineering personnel currently plan to have corrective actions completed by December 2009.

.4 Occupational Radiation Safety

a. Inspection Scope

In addition to the routine review, the inspectors evaluated the effectiveness of PG&E's problem identification and resolution process with respect to the following inspection areas:

- Access Control to Radiologically Significant Areas (Section 2OS1)
- ALARA Planning and Controls (Section 2OS2)

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

.1 Loss of 230 kV Startup Power

a. Inspection Scope

On May 12, 2007, at approximately 10:25 a.m. Pacific Daylight Time (PDT), the offsite startup power was lost to Diablo Canyon Units 1 and 2. The cause of the loss of startup power was due to a transmission line cable that fell from its tower on the Morro Bay-Diablo Canyon 230 kV line. Prior to the event, Unit 1 reactor was defueled and power was supplied by startup power. As a result of the loss of startup power on Unit 1, Diesel Engine Generators (DEGs) 1-1 and 1-2 automatically started and re-energized their respective vital buses per plant design. DEG 1-3 was out-of-service for maintenance; therefore, its vital bus remained de-energized. Since the spent fuel pool cooling pumps are designed to not reload onto their vital buses following re-energization, operators manually restarted the spent fuel pool cooling pumps on their vital buses within five minutes. Spent fuel pool temperature remained at 105°F. Unit 2 reactor remained at 100 percent power throughout the event with the unit's electrical load being supplied by auxiliary power. The Unit 2 DEGs automatically started on the loss of startup power but did not connect onto their vital buses per plant design. Startup power was restored to the site at approximately 11:30 a.m. PDT.

The inspectors responded to the site and observed the operator actions and plant equipment conditions.

b. Findings

No findings of significance were identified.

.2 Unit 1 Manual Reactor Trip in Mode 3 During Control Rod Testing

a. Inspection Scope

On May 27, 2007, operators were performing Surveillance Test Procedure STP R-1C, “Digital Rod Position Indicator Functional Test,” Revision 16, when Control Rod N-13 slipped from 42 steps to 24 steps withdrawn. At the time of the test, Unit 1 was in Mode 3 (Hot Standby). In response to the 18 step deviation and guidance in STP R-1C, operators manually tripped the Unit 1 reactor. Based on review of plant data, industry operating experience, and vendor analysis, PG&E staff concluded that the cause of the rod slippage was due to crud build-up on the control rod drive shaft. The vendor, Westinghouse, recommended that operators exercise Control Bank ‘C’ out and back in five times in order to remove the crud from the drive shaft. During the sequence of five rod exercises, Control Rod N-13 slipped three more times, with each sequential slip occurring at higher steps out of the core (indicating that the crud was moving down and out of the control rod drive mechanism housing). Operators exercised Control Bank ‘C’ five more times without any additional control rod slippage. PG&E staff subsequently concluded that the crud on the Control Rod N-13 drive shaft had been removed to the reactor coolant system.

The inspectors reviewed operator actions and PG&E troubleshooting efforts, as well as equipment performance.

b. Findings

No findings of significance were identified.

.3 (Closed) Licensee Event Report 05000323/200600200, Steam Generator Tube Plugging Due to Stress Corrosion Cracking

On May 19, 2006, PG&E determined that the analysis of eddy current testing on Steam Generator 2-4 indicated that greater than one percent of the tubes were defective as a result of axial outside diameter stress corrosion cracking at the hot leg tube support plates. This determination occurred at the end of Operating Cycle 13. The inspectors verified that PG&E took effective corrective action. All defective tubes were plugged and removed from service in accordance with TS 5.5.9, “Steam Generator (SG) Tube Surveillance Program.” The licensing basis accident assumes a tube plugging limit of 15 percent per steam generator. The plugging percentage for each Unit 1 steam generator remains within the current allowable limit of 15 percent. Steam Generator 1-4 currently has 10.8 percent of its tubes plugged. PG&E maintains a comprehensive program to minimize steam generator tube degradation and plans to replace the steam generators at the end of Operating Cycle 14. This licensee event report is closed.

4OA5 Other

.1 (Discussed) NRC Temporary Instruction 2515/166, PWR Containment Sump Blockage

The inspectors reviewed the Diablo Canyon Unit 1 implementation of plant modifications and procedure changes committed to in their response to Generic Letter 2004-02, “Potential Impact of Debris on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors.”

The inspectors observed fabrication of the new sump strainers prior to being placed inside Unit 1. The inspectors also observed implementation of measures to reduce debris generation and debris transportation during a loss of coolant accident. These measures included modifying doors to reduce the amount of debris transported to the emergency sump, and the installation of devices that reduce the amount of debris carried to the emergency sump. The inspectors also observed portions of the preparation of the site for the new sump strainers. The inspectors also observed portions of the assembly of the replacement strainers on the turbine deck floor in preparation for installation.

During the inspection, PG&E determined that the TS required water level in the refueling water storage tank was not sufficient for the design of the new sump screen. The new design requires full submergence of the sump screen in water during an accident. This full submergence is required to prevent vortexing and air entrainment in the residual heat removal system during a loss-of-coolant accident. PG&E indicated that a license amendment request would be submitted for a new minimum refueling water storage tank level. In the interim, PG&E will be implementing compensatory measures to ensure operability of the residual heat removal system. These measures include placing an administrative requirement on the refueling water storage tank level, revising surveillance procedures to account for the new refueling water storage tank level, and revising the TS bases for the new refueling water storage tank level to determine operability of the residual heat removal system.

PG&E was granted an extension for completion of all measures associated with Generic Letter 2004-02. The extension was based, partially, on PG&E implementing a number of compensatory measures before the December 31, 2007, date given in Generic Letter 2004-02. During the inspection all mitigative actions committed to by PG&E were on schedule to be completed on time.

Final review and acceptance of chemical and downstream effects will be completed by the Office of Nuclear Reactor Regulation. Pending final submittal and acceptance of licensee’s commitments to Generic Letter 2004-02, inspectors will revisit Temporary Instruction 2515/166 for Diablo Canyon Power Plant, Unit 1, at a later date.

40A6 Meetings, Including Exit

Exit Meeting Summary

On April 6, 2007, the inspectors presented the inspection results of the licensed operator requalification inspection to Mr. J. Welsh, Operations Manager, and other members of PG&E’s management staff. PG&E acknowledged the findings presented. The inspectors also asked PG&E whether any materials examined during the inspections should be considered proprietary. No proprietary information was identified. The lead inspector obtained the final biennial examination results and telephonically exited with Mr. J. Bacerra, Licensed Operator Requalification Training Supervisor, on April 16, 2007.

On May 3, 2007, the inspectors presented the occupational radiation safety inspection results to Mr. J. Becker, Station Director, and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

On May 15, 2007, the inspectors presented the results of the inservice inspection and Temporary Instruction 2515/166 inspection to Ms. D. Jacobs, Vice President Nuclear Services, and other members of her staff who acknowledged the findings. The inspectors noted that while proprietary information was reviewed, all such documents had been returned to PG&E, and the information would not be included in this report.

The resident inspection results were presented on July 19, 2007, to Mr. J. Becker, Vice President Diablo Canyon Operations and Station Director, and other members of PG&E management. PG&E acknowledged the findings presented. The inspectors asked PG&E whether any materials examined during the inspection should be considered proprietary. Proprietary information was reviewed by the inspectors and left with PG&E at the end of the inspection.

4OA7 <u>Licensee-Identified Violations</u>

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

- The inspectors reviewed one noncited violation of 10 CFR 20.1602 for failure to maintain control of the access to a posted very high radiation area. Part 20.1602 of Title 10 of the Code of Federal Regulations requires that, in addition to the requirements in Part 20.1601, PG&E shall institute additional measures to ensure that an individual is not able to gain unauthorized or inadvertent access to areas in which radiation levels could be encountered at 500 rads or more in one hour at one meter from a radiation source or any surface through which the radiation penetrates. Contrary to these requirements, PG&E did not maintain constant surveillance and control of the entry to a posted very high radiation area. Specifically, on March 21, 2007, PG&E staff removed an access plug from the 1-1 cation demineralizer cubicle in order to perform maintenance on valve remote operating mechanisms. The doorway from the 1-1 cubicle to other cubicles was posted as a very high radiation area. During periods when no workers were in the 1-1 cubicle, PG&E did not maintain continuous surveillance of access to the posted very high radiation area. The inspectors determined that the finding was of very low safety significance because: (1) it was not an ALARA finding, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. This event was documented in PG&E's corrective action program as AR A0691736.

- 10 CFR 50.55a(a)(3) states, in part, that proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. Contrary to this, PG&E failed to obtain authorization by the Director of the Office Nuclear Reactor Regulation prior to using an alternate method to perform visual examinations and functional testing of snubbers versus the American Society of Mechanical Engineers (ASME) Code, Section XI, requirements identified in 10 CFR 50.55a(g). Specifically, on March 21, 2006, PG&E submitted a relief request to the NRC for their alternate method of snubber examinations and testing as it applied to the 2nd 10-year interval inservice inspection and testing program. However, the 2nd 10-year interval ended for Unit 1 on May 7, 2006, and, for Unit 2, it ended on June 30, 2006. Therefore, for the majority of the 2nd 10-year interval inservice inspection and testing program, PG&E used an alternate method for examining and testing snubbers without prior approval from the NRC. Relief request regarding the alternate method was granted by the NRC for the 2nd 10-year interval on March 29, 2007. Using IMC 0612, Appendix B, the finding was determined not to be suitable for disposition under the Significance Determination Process since it had the potential to impact the NRC's ability to perform its regulatory function. Under the traditional enforcement process, Supplement 2, Section D.5 of the NRC Enforcement Policy describes this finding as a Severity Level IV violation.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

PG&E personnel

J. Bacerra, Licensed Operator Requalification Training Supervisor
J. Becker, Vice President - Diablo Canyon Operations and Station Director
D. Burns, Operations Training Supervisor
J. Haines, Training Manager
R. Hite, Manager, Radiation Protection
D. Jacobs, Vice President - Nuclear Services
S. Ketelsen, Manager, Regulatory Services
K. Langdon, Director, Operations Services
M. Meko, Director, Site Services
K. Peters, Director, Engineering Services
J. Purkis, Director, Maintenance Services
P. Roller, Director, Performance Improvement
D. Taggart, Manager, Quality Verification
R. Waltos, Manager, Emergency Preparedness
J. Welsh, Operations Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000275/2007003-01	NCV	Failure to Scope Reactor Cavity and Containment Structure Sumps Level Indication Into Maintenance Rule (Section 1R12)

Closed

05000323/2006002-03	LER	Steam Generator Tube Plugging Due to Stress Corrosion Cracking (Section 4OA3.4)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment (71111.04)

Action Requests

A0669248 A0674550 A0678472 A0695231 A0695275

Calculations

Number	Title	Revision
M-988	ASW Flows, Temperatures, and Pressures	7

Drawings

Number	Title	Revision
438266	Neutron Detector Positioning Device - Containment Structure	5
438273	Reactor Support - Containment Structure Areas “F” & “G”	5
438274	Reactor Nozzles Area - Containment Structure	4

Procedures

Number	Title	Revision
OP B-2:1	RHR System Alignment Verification for Plant Startup	20
OP E-5:I	Auxiliary Saltwater System - Make Available	29
STP M-26	ASW System Flow Monitoring	27

Miscellaneous Documents

Title	Date/Revision
DCM No. S-17B, “Auxiliary Saltwater System”	18A

Section 1R05: Fire Protection

Procedures

Number	Title	Revision
CP M-6	Fire	29
OM8.ID1	Fire Loss Prevention	18

OM8.ID4	Control of Flammable and Combustible Materials	14
STP M-69A	Monthly Fire Extinguisher Station Inspection Inside the Protected Area	37
STP M-69B	Monthly CO2 Hose Reel and Deluge Valve Inspection	14
STP M-70C	Inspection/Maintenance of Doors	12

Section 1R08: Inservice Inspection Activities

Action Requests

A0695275	A0695946	A0695945	A0695978	A0695981	A0695220
A0651542	A0696400	A0696037	A0696038	A0659349	A0665588
A0695749	A0696018				

Procedures

Number	Title	Revision
NDE PDI-UT-2	Ultrasonic Examination of Austenitic Piping	5
NDE VT-1-1	Visual Examination of Component Surfaces	0
NDE VT-3-2	Visual Examination of Component Interiors	0
NDE PT-1	Visible Dye Penetrant Examination Procedure	3
NDE ET-7	Eddy Current Examination of Steam Generator Tubing	10
STP M-SGTI	Steam Generator Tube Inspection	14
WDI-CAL-002	Pulser/Receiver Linearity Procedure	7
WDI-ET-003	IntraSpect Eddy Current Imaging Procedure for Inspection of Reactor Vessel Head Penetrations	11
WDI-ET-004	IntraSpect Eddy Current Analysis Guidelines	11
WDI-ET-002	IntraSpect Eddy Current Inspection of J-Groove Welds in Vessel Head Penetrations	8
WDI-ET-008	IntraSpect Eddy Current Imaging Procedure for Inspection of Reactor Vessel Head Penetrations with Gap Scanner	8
WDI-UT-010	IntraSpect Ultrasonic Procedure for Inspection of Reactor Vessel Head Penetrations, Time of Flight Ultrasonic, Longitudinal Wave and Shear Wave	13
WDI-UT-013	IntraSpect UT Analysis Guidelines	12
WDI-STD-101	RVHI Vent Tube J-Weld Eddy Current Examination	6

Number	Title	Revision
WDI-STD-114	RVHI Vent Tube ID and CS Wastage Eddy Current Examination	6
WDI-SSP-1036	Reactor Vessel Head Penetration Inspection Tool Operation for Diablo Canyon Unit 1 (PGE)	0

Miscellaneous Documents

Title	Date/Revision
1R14 Steam Generator Degradation Assessment	May 2007

Section 1R11: Licensed Operator Requalification (71111.11)

Procedures

Number	Title	Revision
TQ2.DC3	Licensed Operator, NLO, and Shift Technical Advisor Continuing Training Programs	15
TQ2.ID4	Training Program Implementation	10

Other Items

Scenario, FRC12-A, “ICC/ Degraded Core Cooling”

Scenario, E3ECA33-A, “Steam Generator Tube Rupture”

LORT Simulator Annual Operating Examination (JPMs)

LORT Biennial SRO Written Exam Material

LORT Biennial RO Written Exam Material

Training Program Curriculum Licensed Operator and STA Requalification

Medical Records (10 percent of all licensed operators and a 100 percent sampling of SCBA corrective lenses in Control Room

Curriculum Review Committee Meeting Minutes

Remediation Training Records

Section 1R12: Maintenance Effectiveness (71111.12)

Action Requests

A0668718	A0668719	A0669024	A0675018	A0675433	A0677570
A0696295	A0690152	A0690156	A0584097	A0697144	A0694908
A0693285	A0693330	A0693874	A0694280		

Drawings

Number	Title	Revision
107709, Sheet 2	Safety Injection	40

Procedures

Number	Title	Revision
MA1.ID17	Maintenance Rule Monitoring Program	17
STP V-5C	Emergency Core Cooling System Hot Leg Check Valve Leak Test	27
EOP ECA-3.1	SGTR With Loss of Reactor Coolant - Subcooled Recovery Required	18
OP AP-1	Excessive Reactor Coolant System Leakage	18
OP AP SD-2	Loss of RCS Inventory	16

Section 1R13: Maintenance Risk Assessments and Emergent Work Control (71111.13)

Action Requests

A0692899

Calculations

Number	Title	Revision
PRA06-06	Positive Displacement Pump Allowed Outage Time Extension	1
PRA02-05	Risk Evaluation for Open Vital Breaker Cubicles and Vital Inverters for Seismic	1

Procedures

Number	Title	Revision
AD7.DC6	On-line Maintenance Risk Management	9
OP J-2:VIII	Guidelines for Reliable Transmission Service for DCPP	12
AD4.ID8	Identification and Resolution of Loose, Missing, or Damaged Fasteners	10
AD8.DC51	Outage Safety Management Control of Off-Site Power Supplies to Vital Busses	12A

Work Orders

C0196006

Section 1R15: Operability Evaluations (71111.15)

Action Requests

A0663923	A0692424	A0692494	A0692495	A0692766	A0555584
A0614496	A0693525	A0693647	A0693669	A0697545	A0697605
A0697733					

Drawings

Number	Title	Revision
049258, Sheet 206	Strut 55S-180R	2
102003, Sheet 4	Feedwater System	74
108003	Feedwater System	59

Procedures

Number	Title	Revision
OM7.ID12	Operability Determination	10

Miscellaneous

Title	Date/Revision
ANSI 31.7b - 1971, “1970 Addenda to Nuclear Piping B31.7 - 1969	March 10, 1971
Westinghouse Letter PGE-06-56	May 11, 2006

WCAP-13247 Diablo Canyons 1 & 2 Tavg/Power Coastdown Program Technical Report	August 1992
Calculating Max Leak Limit from JW Expansion Tank	April 19, 2007

Section 1R19: Post-Maintenance Testing (71111.19)

Action Requests

A0692112	A0692114	A0693893	A0693285	A0693330	A0693874
A0699677	A0695752				

Procedures

Number	Title	Revision
STP V-2U4C	Exercising S/G No. 4 Feedwater Isolation and Control Valves	6A
STP V-3P1	Exercising Main Feedwater Regulating Valve and Bypass Valves	28
LT 4-47B	Bypass Feedwater Regulating Valve FCV-1540 Channel Calibration	10
STP M-9A	Diesel Engine Generator Routine Surveillance Test	73A
STP M-9X	Diesel Engine Generator Operability Verification	19
MA2.ID2	Performance Monitoring Equipment Calibration and Usage Control	8
PEP R-3A	Use of Flux Mapping Equipment	4
STP R-22	Thimble Tube Inspection Program	8
STP M-11A	Station Battery and Pilot Cell Condition Monitoring	21
STP M-11B	Station Battery Condition Monitoring	26
STP M-12A	Vital Station Battery Modified Performance Test	15
PMT 03.27	DFWCS Power Ascension Verification Test	0

Work Orders

C0208473	C0207245	C0207665	C0207136

Section 1R20: Refueling and Other Outage Activities

Procedures

Number	Title	Revision
MA1.ID14	Plant Crane Operating Restrictions	14
MP M-7.1A	Reactor Vessel Closure Head Removal	4
OP A-2:II	Reactor Vessel - Draining the RCS to the Vessel Flange - With Fuel in Vessel	31

Section 1R22: Surveillance Testing

Action Requests

A0641000	A0655759	A0697715	A0694888	A0695249

Procedures

Number	Title	Revision
STP R-10C	Reactor Coolant System Water Inventory Balance	32
STP R-10E	RCS Leakage Step Increase Evaluation	0
STP I-1B	Routine Daily Checks Required by Licenses	82
STP M-15	Integrated Test of Engineered Safeguards and Diesel Generators	39
STP V-600	General Containment Isolation Valve Leak Tests	21
STP V-630	Penetration 30 Containment Isolation Valve Leak Testing	23

Miscellaneous

Title	Date/Revision
Letter from David Terao, NRC, to John Keenan, PG&E, “Diablo Canyon Power Plant, Unit Nos. 1 and 2 - Relief Request NDE-SBR for the Second 10- Year Interval Inservice Inspection and Examination Program for Snubbers (TAC Nos. MD0535 and MD0536)”	March 29, 2007

Section 2OS1: Access Controls to Radiologically Significant Areas (71121.01)

Action Requests

A0674391	A0674761	A0675104	A0675525	A0678653	A0679778
A0686985	A0689069	A0694205	A0694258	A0694685	

Audits and Self-Assessments

Quality Verification Assessment 070040059, Review of Rapid Containment Entry Process
Quality Performance Assessment Report, 1st Period 2006
Quality Performance Assessment Report, 2nd Period 2006
Quality Performance Assessment Report, 3rd Period 2006
Quality Performance Assessment Report, 4th Period 2006

Radiation Work Permits

SWP 1001	1R14 General Access to Containment
SWP 1002	1R14 Scaffolding in Containment
SWP 1015	1R14 Minor Work in HRA/LHRA/VHRA in Containment
SWP 1027	1R14 Reactor Reassembly

Procedures

Number	Title	Revision
RP1	Radiation Protection	4A
RP1.DC4	Radiological Hot Spot Identification and Control Program	2
RCP D-215	Radiological Coverage of Underwater Work	5
RCP D-220	Control of Access to High, Locked High, and Very High Radiation Areas	32
RCP D-222	Radiation Protection Lock and Key Control	5
RCP D-230	Radiological Control for Containment Entry	17
RCP D-420	Sampling and Measurement of Airborne Radioactivity	18A
RCP D-430	Plant Airborne Radioactivity Surveillance	16
RCP D-500	Routine and Job Coverage Surveys	23

Section 4OA2: Problem Identification and Resolution (71152)

Action Requests

A0668922	A0668929	A0669222	A0669227	A0669270	A0669468
A0674806	A0696833	A0698847			

Miscellaneous

Title	Date/Revision
STP V-5A2, “Emergency Core Cooling System Check Valve Leak Test, Post-Refueling/Post-Maintenance Valves 8948 A-D, 8818 A-D, and 8819 A-D”	18

Section 4OA3: Followup of Events and Notices of Enforcement Discretion

Action Requests

A0654144	A0699025	A0699045	A0565847	A0699162	A0699496

Miscellaneous

Title	Date/Revision
Event Notification 43391, “Manual Reactor Trip in Mode 3 During Control Rod Testing	May 27, 2007
Event Notification 43393, “Emergency Diesel Generator Actuation Due to Momentary Undervoltage Condition	May 28, 2007
OP E-4:I, “Circulating Water System - Prepare for Service”	June 1, 2007

Section 4OA5: Other

Calculations

Number	Title	Revision
N-042	Fibrous Material Debris and Calcium Silicate Insulation Vapor Barrier Debris From HELB Inside Containment	12
N-100	Maximum Flow From ECCS Pumps and Minimum Flow to Containment Spray Header	2
M-227	Post LOCA Minimum Containment Sump Level	4
M-591	Determine the Head Loss Across the Recirculation Sump Screen Structures	32
M-1093	Diablo Canyon Unit 1 Chemical Effects Debris Calculation	1

Evaluations

Number	Title	Revision
WES007-PR-02	Evaluation of Containment Recirculation Sump Upstream Effects for the Diablo Canyon Power Plant	0

Design Change Packages

Number	Title	Revision
A0672569	Modify Door 277 to Install Debris Interceptor	1
A0679235	Modify Doors 275 and 276 in Unit 1 Containment Structure to Install Debris Interceptors During 1R14	0
A0671528	Modify Unit 1 Reactor Cavity Door No. 278	0
C-49857	Installation of a Larger Sump Screen	1

LIST OF ACRONYMS

ADAMS	agency document and management system
AFW	auxiliary feedwater
ALARA	As Low As is Reasonably Achievable
AR	action request
ASME	American Society of Mechanical Engineers
CCW	component cooling water
CFCUs	containment fan cooler units
CFR	*Code of Federal Regulations*
DEG	Diesel Engine Generator
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
FSAR	Final Safety Analysis Report
IMC	Inspection Manual Chapter
LER	Licensee Event Report
NCV	noncited violation
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PG&E	Pacific Gas and Electric Company
PI	Performance Indicator
RCS	reactor coolant system
RHR	residual heat removal
SDP	Significance Determination Process
SLUR	second-level undervoltage relays
TS	Technical Specifications